November 18, 1999

Mr. R. P. Necci, Vice President
Nuclear Oversight and Regulatory Affairs
c/o Mr. D. A. Smith, Manager - Regulatory Affairs
Northeast Nuclear Energy Company
P.O. Box 128
Waterford, Connecticut 06385

SUBJECT: NRC INSPECTION REPORT 50-336/99-10 AND 50-423/99-10

Dear Mr. Necci:

This refers to the NRC Engineering Team Inspection that was conducted at the Millstone Nuclear Generating Station, from September 7-10, 1999, September 20-24, 1999, and October 21-22, 1999. The overall objective of the inspection was to evaluate engineering support to safe plant operation. The inspection was directed toward areas important to public health and safety. The areas examined during this inspection included engineering performance in plant modifications, technical issue identification and resolution, and safety evaluations. At the conclusion of the inspection, the preliminary inspection findings were discussed with members of your staff.

Overall, we found that engineering provided good support to plant operations and maintenance. We noted the plant modifications were properly designed and implemented, and the modification documents were of good quality. In general, the completed safety evaluations were comprehensive and thorough. We also determined that engineering was effective in identifying and properly resolving technical issues.

Based on the results of our inspection, we have determined that a violation of NRC requirements occurred regarding corrective actions for identified problems that were not always timely or effective. Our inspection determined that your Independent Safety Evaluation Group reviews, Engineering Oversight Assessments, and Engineering Self-Assessments identified instances where previously identified deficiencies had either not been corrected, or were not corrected in a timely manner. A specific example was the identified deficiency associated with the training documentation of engineering personnel that was not corrected in a timely manner to prevent a recurrence of the problem. This violation is being treated as a Non-Cited Violation (NCV), consistent with Appendix C of the Enforcement Policy. This NCV is described in the subject inspection report. You do not need to respond to the violation, but you may contest the violation or severity level. If you so choose, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Millstone facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be placed in the NRC Public Document Room (PDR).

We appreciate your cooperation.

Sincerely,

ORIGINAL SIGNED BY:

Lawrence T. Doerflein, Chief
Engineering Programs Branch
Division of Reactor Safety

Docket No. 50-423

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U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket Nos.:	50-336 50-423
License No.:	NPF-49
Report No.:	99-10
Licensee:	Northeast Nuclear Energy Company P. O. Box 128 Waterford, CT 06385
Facility:	Millstone Nuclear Power Station, Unit 3
Inspection at:	Waterford, CT
Dates:	September 7-10, 1999, September 20-24, 1999 and October 21-22, 1999
Inspectors:	R. L. Fuhrmeister, Sr. Reactor Engineer, Engineering Programs Branch S. K. Chaudhary, Sr. Reactor Engineer, Engineering Programs Branch T. F. Burns, Reactor Engineer, Engineering Support Branch R. S. Bhatia, Reactor Engineer, Engineering Support Branch K. M. Jenison, Project Engineer, Millstone Inspection Directorate (MID)
Approved by:	Lawrence T. Doerflein, Chief Engineering Programs Branch Division of Reactor Safety Region I

EXECUTIVE SUMMARY

Millstone Nuclear Generating Station, Units 2&3
NRC Inspection Report 50-336/99-10 and 50-423/99-10

Introduction

An on-site team inspection of the Engineering area was conducted at the Millstone Nuclear Power Station, during the period of September 7-10, September 20-24 and October 21-22, 1999. The overall objective of the inspection was to determine whether engineering provided proper support for safe operation of the facility. The inspection included 1) an evaluation of the 10 CFR 50.59 safety evaluation program related to changes, tests, or experiments at the plant, and 2) a comparison of operability determinations, safety evaluations, engineering work requests and condition report engineering dispositions between Unit 2 and Unit 3 for the six months preceding the inspection.

The team focused on systems with high risk significance. The target systems for the inspection were selected from a list of the ten most significant systems, avoiding those systems reviewed by the Operational Safety Team Inspection. The systems selected were auxiliary feedwater, 4160VAC electrical distribution, 125VDC electrical distribution, service water, feedwater, and refueling water storage tank.

Engineering

- Design change modifications were properly designed and implemented. The safety evaluations provided sufficient bases to demonstrate that no unreviewed safety questions were involved in the modifications. The design change documents were well written and thorough. The post modification tests were conducted appropriately to test the systems prior to declaring the system operable. The supporting calculations, where applicable, were appropriate to justify the design changes. Affected documents were appropriately updated to reflect the design changes. (Section E1.1)

- Temporary modifications were properly designed and implemented. The design changes correctly addressed the concerns for which the modifications had been developed. The evaluation, installation, post-modification-test requirements, and safety reviews provided by engineering presented adequate technical basis for the modifications. (Section E1.2)

- Overall, the sampled Operability Determinations (OD), Safety Evaluations (SE), Design Change Request (DCR) summaries, root cause analyses and 10 CFR 50.59 assessments were found to be acceptable, adequately managed and timely. Technical discussions were well written, adequately supported and thoroughly referenced. Applicable root cause analyses were detailed and the depth of the analyses was commensurate with the significance of the issues. When compared to Unit 2, Unit 3 ODs, 10 CFR 50.59 assessments, root cause evaluations, DCR Summaries and SEs

were found to be more detailed, more clearly written and better supported by reference. These differences contributed to the perception of a difference in overall review quality between the two units, although, no significant difference in engineering support was identified. (Section E2.1)

- Discrepancies in the calculation for the Unit 3 containment sump trisodium phosphate concentration used to support the Technical Specification requirements resulted in an unresolved item (URI 50-423/99-10-01). (Section E2.1)

- Based on the quality of the selected Engineering department work products and training/qualification documentation supplied from various sources, the team concluded that the sampled Engineering department engineers and supervisors who provided technical information, analysis and operational support possessed adequate technical expertise to provide the engineering service. (Section E4.1)

- The Independent Safety Engineering Group (ISEG) provided valuable feedback to the line organizations regarding the operation of the Millstone facility. The ISEG had been effective in identifying deficiencies and areas for improvement. The ISEG also determined that corrective actions for some prior deficiencies were ineffective or untimely. A specific example was the identified deficiency associated with the training documentation of engineering personnel that was not corrected in a timely manner to prevent a recurrence of the problem. The failure to correct conditions adverse to quality in a manner to prevent recurrence was a non-cited violation (NCV 50-423/99-10-02). (Section E7.1)

- In general, NNECO's corrective action process, the Condition Report (CR) program, adequately controlled, tracked, identified, resolved, and prevented recurrence of problems identified in Engineering department self assessments, Operability Determinations, Design Change Report summaries, Safety Evaluations and Nuclear Oversight reviews. No significant difference in overall CR program implementation quality was identified between the two units. (Section E7.2)

- NNECO Nuclear Oversight adequately performed assessments of Engineering department related activities. Assessment activities and findings were appropriately oriented toward maintaining the design basis of the units, ensuring the operability of safety related equipment, identifying conditions of increased risk, and maintaining NNECO quality standards. Most Nuclear Oversight assessment findings were administrative and/or process related in nature and the most significant assessment findings were generally related to inadequate corrective actions for a previously identified CR condition. (Section E7.3)

- NNECO Engineering department self assessments were detailed, varied in scope, well supported, thoroughly referenced and safety oriented. Many of the self assessment findings were of a technical nature, but most were administrative. Assessment findings were assigned to the CR process, given appropriate CR significance, adequately tracked and managed, and corrected in a timely manner. Typical self assessment issues were assigned CR significance levels of 2 or 3, and had no immediate impact on the safe operations of the units. (Section E7.4)

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Report Details

I. ENGINEERING

E1 Conduct of Engineering

E1.1 Plant Design Modifications

a. Inspection Scope (37550)

The team reviewed the preparation and implementation of selected permanent plant design modifications installed during the last refueling outage to verify the design change process complied with the applicable plant administrative procedures and regulatory requirements, and to evaluate the effectiveness of Northeast Nuclear Energy Company's (NNECo's) engineering activities in ensuring the plant design bases were maintained, and operational safety was assured. The team also conducted walkdowns of selected installations to verify their conformance with applicable documents.

b. Observations and Findings

Permanent plant design changes at the Millstone Nuclear Power Station were governed by design control manual DCM-03. This manual outlined the process by which design change records and minor modifications were required to be prepared, reviewed, approved, implemented and closed. The team found the established modification procedures contained adequate detail to provide appropriate guidance for the engineering organization to evaluate, develop, and document the essential elements of the design changes.

Overall, the modifications were well designed and documented. The team found that the design change packages contained detailed design and installation requirements; including the bases of the current design, method of change, design inputs, and adequate procurement requirements. With the exception noted below, the packages included proper safety evaluations, implementation and operational considerations, and the post modification test requirements. The team determined that installation instructions were detailed with appropriate acceptance tests and acceptance criteria clearly specified. The team found that the applicable design documents, procedures, and calculations had been updated to reflect the design changes, as required by the design control procedures. The team reviewed a portion of the applicable drawing revisions and vendor manual changes and determined the modifications had been captured into the controlled documents system. A walkdown of selected installations revealed no problems.

One modification, M3-98-039, Service Water Piping Modification, was scheduled for installation during the last refueling outage; however, the implementation of the design change had been delayed. This modification involved the installation of a pair of flanges on each ten inch line that connects the service water system to the auxiliary feedwater system to facilitate spool piece removal. The spool pieces were intended to allow internal inspections that were required to meet commitments made in response to NRC Generic Letter 89-13, and to detect erosion, corrosion, and biofouling. Currently, NNECo is in the process of re-evaluating the necessity of this proposed modification.

DCR M3-99004, Auxiliary Feedwater Pump Design Changes to 3FWA*P2

The governor valve for the Auxiliary Feed Water (AFW) pump turbine had a poor operational history due to binding attributed to corrosion of the governor valve stem. The review of safety evaluation S3-EV-99-0009, which supported the modification, revealed a weakness in the technical basis for the replacement of the governor valve stem base material. The replacement material was recommended by the valve vendor who indicated the material change would increase corrosion resistance of the stem. The safety evaluation performed by the licensee provided no technical discussion regarding the suitability of this material for service other than reference to the vendor recommendation as “suitable in form, fit and function”. The licensee provided further that “the material change can only improve reliability and reduce the probability and consequences of equipment malfunction and/or accidents previously evaluated in the SAR [safety analysis report]”.

The team noted during review of the modification package and interviews of licensee personnel that the vendor had made previous recommended material changes for this application which proved to be unsatisfactory. Interviews with licensee personnel revealed that there were early efforts to produce an “equivalency evaluation” of this specific stem material change recommended by the vendor to confirm expectations that this material (Inconel 718) would be suitable for service. This effort was not formally concluded and therefore was not considered a basis to support the material change.

Installation of the replacement material at other nuclear facilities revealed that this stem material was susceptible to “binding” with the surrounding steel washers and carbon spacers, as identified in NRC Information Notice 98-24, “Stem Binging in Turbine Governor Valves in Reactor Core Isolation Cooling (RCIC) and Auxiliary Feedwater (AFW) Systems.” The binding was attributed to thermal expansion of the stem upon heat up to component operating temperature as there had been no compensating change in the spacer hole size. The replacement material exhibited a thermal expansion rate considerably in excess (40%) of prior valve stem materials (type 410 stainless steel). The stem, washers and carbon spacers were not installed at the Millstone site until after the material thermal expansion problem had been identified and corrected by the Vendor (10 CFR 21 report, April 16, 1998). The team found that the vendor had identified, as part of its 10 CFR 21 investigation, that the valve stem binding problems which were described in the information notice all occurred with valve stems manufactured by a third party, to incorrect dimensions provided by another utility from reverse-engineering a spare stem. Since NNECo purchased their valve stem from the original equipment vendor, and the certificate of compliance provided with the stem specifically stated that the parts “...are compatible and interchangeable with their original equipment...” the team concluded that the lack of a specific evaluation of material properties was acceptable.

c. Conclusions

Overall, design change modifications were properly designed and implemented. The safety evaluations provided sufficient bases to demonstrate that no unreviewed safety questions were involved in the modifications. The design change documents were well written and thorough. The post modification tests were conducted appropriately to test the system prior to declaring the system operable. The supporting calculations, where applicable, were appropriate to justify the design changes. Affected documents were appropriately updated to reflect the design changes.

E1.2 Temporary Modifications

a. Inspection Scope (37550)

The team reviewed a sample of temporary modifications (TM) to determine whether the TMs were properly designed and implemented in compliance with plant established administrative procedures and regulatory requirements. The TMs were also reviewed to determine the extent of engineering involvement, quality of design inputs, implementation of safety evaluations, and post installation testing requirements.

b. Observations and Findings

The team reviewed several TMs, including both open and closed modifications. The TMs reviewed were found to be prepared and implemented in compliance with the licensee’s procedure (WC-10, Temporary Modifications) which governed the TM process. The team reviewed each TM to determine if the required safety evaluation screening process had been completed and when required, the safety evaluation was completed, reviewed and approved. The safety evaluations were reviewed to determine the extent and quality of engineering involvement. The team found the safety evaluation screening process provided a logical approach with good technical bases for the conclusions. The team reviewed the completed safety evaluations and found them to provide adequate analysis of the affect on plant safety and licensing requirements. The team noted that the modifications had been prepared with good engineering involvement in providing design inputs, technical evaluations and specification of installation and post work testing requirements.

The team observed that the modification packages were adequately prepared and documented, and conformed to the station’s procedural requirements. The modification packages included relevant drawings, safety evaluations, PORC approvals, and the installation verification instructions for the TMs.

The station had fourteen outstanding TMs at the time of this inspection, with a goal of ten. At the time of the inspection, the number of TMs installed in the plant was trending down.

c. Conclusions

The temporary modifications were properly designed and implemented. The design changes correctly addressed the concerns for which the modifications had been developed. The evaluation, installation instructions, and safety reviews provided by engineering presented adequate technical bases for the modifications.

E2 Engineering Support of Facilities and Equipment

E2.1 Operability Determinations, Safety Evaluations, Design Change Request (DCR) Summaries and 10 CFR 50.59 Assessments

a. Inspection Scope (37550, 92903)

The team reviewed a sample of Operability Determinations (ODs), Safety Evaluations (SEs), Design Change Request (DCR) summaries and 10 CFR 50.59 Assessments, to assess the adequacy of engineering involvement in and support for the resolution of operating, technical and regulatory issues.

b. Findings and Observations

Overall Quality of ODs, SEs, 50.59 Assessments and DCR Summaries

Based on the sample of ODs, SEs, DCR summaries, root cause evaluations and 10 CFR 50.59 assessments, the team found the resolution of the selected issues to be technically adequate, generally well written and documented, technically well supported, thoroughly referenced, and completed in a timely manner. Issues were adequately administered and tracked in accordance with NNECO procedures. Operability was routinely discussed and evaluated in terms of the ability of the components to perform their safety functions and within the context of the design and licensing bases. Root cause analyses were usually detailed and were accomplished at a depth commensurate with the safety significance of the issues. Followup actions were delineated as Condition Reports (CRs), entered into the corrective action process, and assigned appropriate levels of significance. Based on the sampled activities, engineering support for ODs, SEs, DCRs and 10 CFR 50.59 assessments was determined to be adequate.

Comparison of Work Products Between Units

The Engineering department support activities of both units were based on the same reference NNECO procedures and policies. When compared to Unit 2, the Unit 3 ODs, 10 CFR 50.59 assessments, DCR summaries, root cause evaluations, and SEs were found to be more detailed, more clearly written and better supported by reference. These differences contributed to the appearance of a difference in overall analysis quality between the two units, although no significant difference in engineering support, technical content or work product quality was identified by the team.

Safety Evaluation, S3-EV-99-0009 - Auxiliary Feedwater Pump Rotating Assembly and Governor Valve Stem Material Replacement

S3-EV-99-0009 addressed, among other things, a replacement stem for the turbine-driven auxiliary feedwater pump governor valve, 3MSS-MCV5. The material for the stem was changed from a nitrided 410 stainless steel to an Inconel 718 alloy. The design change associated with this SE was discussed in Section 1.1 of this report. The assessment did not address thermal expansion or other mechanical properties of the new stem which were critical attributes and differed from those of the replaced stem. The safety evaluation relied on an assertion of "fit, form, and function," from the vendor. As discussed in Section 1.1, stem binding issues were identified and resolved prior to purchase of the parts for Millstone 3, and the vendor certificate of compliance stated the parts were compatible and interchangeable with the original equipment.

Safety Evaluation,E3-EV-98-0006 - Auxiliary Feedwater (AFW) Flow Following a Steam Generator Tube Rupture (SGTR)

The team identified what appeared to be an incorrect post accident flow assumption in E3-EV-98-0006, concerning motor driven AFW flow following a SGTR. The licensee provided the team with DCR summary M3-98-007 and Engineering Work Request (EWR) 96-0389. The DCR summary amended the SE assumption and concluded that AFW flow was not evenly divided between all four steam generators following a SGTR, but that the assumption was conservative from the standpoint of offsite dose. The DCR summary evaluated the impact of the assumption on the associated post SGTR heat transfer model and found it to not be significant. This issue was adequately resolved by the licensee prior to the team's finding and no violation of NRC requirements was identified.

<u>DCR Summary M3-97083 - Setpoint Revision for Reactor Coolant Pump Underspeed Reactor Trip</u>

The team identified what appeared to be an incorrect assumption, concerning reactor coolant pump coast down rates following an FSAR, chapter 15, design basis, Loss of Flow transient. The DCR summary concluded that two low frequency cases were bounded by the equilibrium 60 Hz case discussed in the FSAR. Subsequent to the writing of this DCR summary and prior to the team identifying this issue the licensee documented a grid frequency variance problem in its CR process and resolved the problem through an FSAR update. No violation of NRC requirements was identified.

<u>Evaluation and Approval of US(B)-350, Unit 3 Containment Sump Trisodium Phosphate (TSP) Concentration and Iodine Partition Coefficient Calculations</u>

The team reviewed the calculations, analyses and operational controls used by the licensee to establish and maintain the amount of TSP needed to keep the containment sump in a desired pH range following a Loss of Coolant Accident (LOCA). Several observations were made by the team concerning these activities:

- Section (2) of the calculations stated that long term containment sump pH of 7.1 was specified for the determination of the amount of TSP to be installed inside the containment and that the calculations were to include assumptions 1 through 4. Assumption 1 stated that all boric acid liquid sources would be dumped without considering spray hold up or delay. The team determined that the calculations did not document a treatment of the boric acid storage tanks, the volume control tank, or the pressurizer relief tank volumes. Based on conversations with NNECO engineers, NNECO determined that an assumption of 2900 ppm for RCS boric acid concentration was conservative. It was not clear that this assumption was validated in the calculations.

- Assumption 4 stated that the calculation was to establish the TSP basket volumes. Step 2 on page 14 of the calculation assumed a cross sectional area for the twelve baskets without reference or validation. In addition, the calculation failed to verify that the fill line was appropriately marked to account for the desired volume of TSP. Subsequent to this calculation and prior to the team's observation, the licensee identified and analyzed two related problems with TSP basket volumes. The first problem was that the mixture settled and dropped below the fill level. The second problem was that baskets were overfilled following the licensee discovery that the material had settled.

- Calculation Input Section 3, item 1 assumed an RCS LOCA starting temperature of 583.5 degrees F, which approximately correlates to a 100% full power condition in the RCS. This assumption is not conservative with respect to maximizing the density of RCS coolant and its impact on containment sump pH. The team determined that the maximum RCS coolant density should occur at about 0% power or approximately 557 degrees F.

- There was an assumption included in the TSP titration analysis that stated a pH of 7.13 was needed to ensure that a pH of 7.10 was maintained. Aside from this statement there was no treatment or statistical analysis of measurement error or variation in the calculation on the mass requirement of TSP.

- The pH calculations addressed HCl and HNO_3 post-LOCA contributions, but did not address pH contributions from other sources in containment, such as: emulsified coatings (other than top coating); concrete degradation, dust or damage; corrosion materials; insulation materials; or other chemicals inside containment.

The team identified five discrepancies in calculation US(B)-350, Unit 3 Containment Sump Trisodium-phosphate (TSP) Concentration and Iodine Partition Coefficient, which supports the Millstone Unit 3 Technical Specification (TS) requirements for TSP volume. This issue is unresolved, pending review of additional information from the licensee to resolve the five apparent discrepancies and ensure that the TS requirements have adequate supporting calculations for the bases. **(URI 50-423/99-10-01)**

c. Conclusions

Overall, the sampled Operability Determinations (OD), Safety Evaluations (SE), Design Change Request (DCR) summaries, root cause analyses and 10 CFR 50.59 assessments were found to be acceptable, adequately managed and timely. Technical discussions were well written, adequately supported and thoroughly referenced. Applicable root cause analyses were detailed and the depth of the analyses was commensurate with the significance of the issues. When compared to Unit 2, Unit 3 ODs, 10 CFR 50.59 assessments, root cause evaluations, DCR Summaries and SEs were found to be more detailed, more clearly written and better supported by reference, although, no significant difference in quality of engineering support was identified.

Discrepancies in the calculation for the Unit 3 containment sump trisodium phosphate concentration used to support the technical specification requirements resulted in an unresolved item **(URI 50-423/99-10-01)**.

E4 Engineering Staff Knowledge and Performance

E4.1 Engineering Knowledge and Performance

a. Inspection Scope (93802, 37550)

Based on a licensee developed list of risk significant systems, the team conducted system reviews; inspected selected training records, qualification standards and summaries; evaluated NNECO Nuclear Oversight audits of Engineering department activities including training and qualification, and reviewed selected Engineering department self assessments. The reviews performed by the team were intended to evaluate the actual experience and training of selected responsible engineers and supervisors with technical approval responsibility, and to assess their effectiveness in assuring the design basis and operability of risk significant systems. The team also evaluated the coordination, tracking and administration of Engineering department training/qualification records to determine if the current NNECO qualification record management procedure (TQ-1, Personnel Training Qualification) was being adequately implemented.

b. Observations and Findings

The team selected several systems that appeared in recent CRs as a result of specific events or system performance. Documentation reviews included selected Operability Determinations (ODs), Safety Evaluations (SEs), root cause analyses, Special Procedures (SPROC), Design Change Request (DCR) summaries and 10 CFR 50.59 Assessments. The team's review found that the events had been well documented, and thoroughly evaluated, including both equipment and human performance considerations. Based on the quality of the selected Engineering department work products and training/qualification documentation supplied from various sources, the team concluded that the engineers supplying the technical information and operational support possessed the technical competency to perform the engineering service. The training documentation reviewed by the team included various descriptions of technical experience, training, qualification, professional education, and professional certification.

The team reviewed the technical qualifications of a sample of engineering supervisors who had technical approval responsibility. The team determined that the sampled supervisors possessed an adequate depth of technical experience, training, qualification, and professional education; were responsive to site operational and regulatory needs; were directly involved in troubleshooting and/or coordinating the resolution of technical issues; and routinely interfaced with other departments during the resolution of technical issues. Based on the resolution of specific issues, the team determined that interfaces among engineering groups were good and the communication between engineering and operations was generally effective.

The team found that Engineering qualification and training records were administered and managed in a diverse manner, depending on the engineering organization (section or group) and/or the type of training/qualification record being maintained. Summaries for some records were located in the Northeast Utilities Training Information Management System (NUTIMS). Training and qualification records were also located in the Nuclear Document Services system, the Nuclear Training department and local Engineering department supervisor files.

The team determined that the quality of engineering support to plant operations, within the management control processes established by NNECO, was dependent on the application of appropriately qualified and trained engineers, independent reviewers, and supervisors who were responsible for technical approval of engineering products. In the case of ODs the ultimate authority for determining operability was the responsibility of the Operations Shift Manager (OSM). However, in many cases the technical engineering support supplied to the OD was outside/beyond the immediate expertise of the OSM and is dependent on the technical competency of the Engineering department personnel supplying the support. To ensure the quality of engineering support NNECO established a process of task and process qualification which was dependent on specific training. Procedure TQ-1, “Personnel Training Qualification,” established the methods and means to document, store and tabulate professional training in the Engineering department.

The team reviewed a sample of the ODs and SEs referenced in CRs M2-99-0683, M2-99-0684, and M3-99-2575, to determine if the identified training record deficiencies (see section E7.1) affected the technical quality of the analyses or work product. The team was not able to identify any technical or work product errors that were related to the identified deficiencies in the Engineering department training and qualification records. A sample of Engineering department self assessments was also reviewed by the team. None of the Engineering department self assessments identified technical or work product errors related to the failure to fully implement TQ-1. The team interviewed two members of the ISEG task group that performed the audit which produced CRs M2-99-0683, M2-99-0684, and M3-99-2575, in order to determine if the ISEG had identified training record deficiencies that affected the technical quality of an analysis or work product. Based on discussions with the ISEG, the team determined that a technical product review had not been an attribute of the ISEG audit, and the ISEG task group had formed no opinion concerning the technical quality of the SEs identified in its report. The team concluded that there was no identified connection between the failure to fully implement TQ-1 and the technical quality of the analyses and/or other supporting work products.

c. Conclusions

Based on the quality of the selected Engineering department work products and training/qualification documentation supplied from various sources, the team concluded that the sampled Engineering department engineers and supervisors who provided technical information, analysis and operational support possessed adequate technical expertise to provide the engineering service.

E7 Quality Assurance in Engineering Activities

E7.1 Independent Safety Engineering Group

a. Inspection Scope (37550)

The team reviewed Independent Safety Engineering Group (ISEG) reports which had been issued during the past year to evaluate the scope of reviews and the significance of the findings, reviewed technical specifications relating to ISEG, and discussed line organization response to ISEG findings with several members of the engineering organization.

b. Observations and Findings

The ISEG function was fulfilled by a group of individuals within Nuclear Safety Engineering (NSE) for Millstone Unit 3. The Unit 3 Technical Specifications, Section 6.2.3.1 also assigns the function of operating experience reviews to the ISEG. ISEG activities are governed by procedure NOQP 3.04, Revision 3, “Nuclear Safety Engineering Group Functions and Responsibilities - Independent Safety Engineering Group and Operating Experience Assessment.”

The ISEG activities described in the reports reviewed were a combination of preplanned, requested, and spontaneous observations of activities at the facility. The activities included engineering work, conduct of operations, and maintenance.

Independent Safety Engineering Group (ISEG) report, AR 98020656, “Engineering Qualification,” dated February 9, 1999, and associated Condition Reports M2-99-0683, and 0684 identified that the training and qualification records for some Engineering department personnel, who performed Safety Evaluations (SEs) and Operability Determination (ODs), did not meet the requirements of procedure TQ-1, “Personnel Training and Qualification.” In addition, the CRs and ISEG report identified ten CRs and an Engineering department self assessment that had previously identified similar training record quality problems. Subsequent to the closure of CRs M2-99-0683 and 0684, Engineering Department self assessments and numerous additional CRs continued to identify training and qualification record problems, and the failure to fully implement TQ-1. On July 7, 1999, the ISEG initiated CR M3-2575, to re-address the failure to fully implement TQ-1. On September 22, 1999, the Millstone Management Review Team (MRT) reviewed CR M3-99-2575 and initiated corrective actions to address the training records problems initially identified by the ISEG in February 1999.

The ISEG reports reviewed provided valuable insights into the work processes, included recommendations for improvement, and identified a number of deficiencies. The most significant deficiencies identified by ISEG were related to previously identified problems which had not been adequately corrected, or, in some instances, corrected in a timely manner. A specific example was the identified deficiency associated with the training documentation of engineering personnel that was not corrected in a timely manner to prevent a recurrence of the problem. Therefore, NNECO failed to implement the requirements of procedure TQ-1, Personnel Training Qualification. Other deficiencies that NNECO identified that were not properly corrected was the adverse trend in reactivity management and control and outage Risk Management weakness from 3RFO5 which recurred during 3RFO6 (schedule did not contain a section to identify critical safety systems and structures that must be available to maintain defense in depth, some prerequisites required by procedures OM-1 and OM-2 had not been performed).

The failure to adequately respond to and correct conditions adverse to quality in a manner to prevent recurrence was a violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Actions". This Severity Level IV violation was treated as a Non-Cited Violation, consistent with Appendix C of the NRC Enforcement Policy, these issues are in the licensee's corrective action program as CR M3-99-2575, CR M2-99-0776 and CR M3-99-1624. **(NCV 50-336&423/99-10-02)**

c. Conclusions

The team concluded that the ISEG provided valuable feedback to the line organizations regarding the operation of the Millstone facility. The ISEG had been effective in identifying deficiencies and areas for improvement. The ISEG also determined that corrective actions for some prior deficiencies were ineffective or untimely. A specific example was the identified deficiency associated with the training documentation of engineering personnel that was not corrected in a timely manner to prevent a recurrence of the problem. The failure to correct conditions adverse to quality in a manner to prevent recurrence was a non-cited violation.

E7.2 Problem Identification and Resolution

a. Inspection Scope (40500)

The team conducted a performance based evaluation to determine the effectiveness of the licensee's Condition Report (CR) program in identifying, resolving, and preventing problems that were identified in Engineering department self assessments, ODs, DCR summaries, SEs and Nuclear Oversight assessments.

b. Observations and Findings

The team determined that the assignments of the CR significance levels and classifications were generally appropriate and in accordance with the CR program guidance. More significant issues received detailed reviews, including a root cause analysis, while those of lower significance received less investigation and on occasion administrative closure.

The team found communications between the Operations departments, the Engineering department and the Management Review Team (MRT), regarding CR issues, to be adequate. New CR issues were presented at the daily MRT meetings. The issues were classified by the responsible department managers, and discussion of the CR topics was usually good. The CR performance indicators provided a good overview of CR program effectiveness, which appeared to be acceptable. The team determined that in general, the licensee's CR program adequately identified, resolved, tracked and prevented problems that were identified in a selected sample of Engineering department self assessments, ODs, DCR summaries, SEs and Nuclear Oversight assessments. An exception to this general finding was the resolution of the CRs related to ISEG report AR

98020656, which is discussed in section E7.1 of this report. The team noted that although the ISEG was aggressive in the pursuit of resolution of issues identified in AR 98020656, it did not fully employ the conflict resolution processes that were available to it, including one process that was internal to the Nuclear Oversight organization.

c. Conclusions

In general, NNECO's corrective action process, the Condition Report (CR) program, adequately controlled, tracked, identified, resolved, and prevented recurrence of problems identified in Engineering department self assessments, Operability Determinations, Design Change Report summaries, Safety Evaluations and Nuclear Oversight reviews. No significant difference in overall CR program implementation quality was identified between the two units by the NRC or NNECO Oversight.

E7.3 Nuclear Oversight of Engineering

a. Inspection Scope (37550)

The team reviewed a sample of Nuclear Oversight assessments to determine the adequacy of Nuclear Oversight's involvement in and support for the resolution of operating, technical and regulatory issues associated with Millstone Engineering department activities. Oversight assessments were evaluated to determine Nuclear Oversight's contribution towards maintaining the design basis of the units, ensuring the operability of safety related equipment, establishing conditions of reduced risk, and maintaining NNECO quality standards.

b. Observations and Findings

Based on a sample of Nuclear Oversight assessment activities, the team determined that Nuclear Oversight assessments were detailed, oriented towards those systems/ processes with the greatest impact on safety and risk, and maintained NNECO quality standards. Most of the findings and CRs that resulted from the sampled Nuclear Oversight assessments were administrative and/or process related. Assessment findings were assigned appropriate CR significance, and corrective actions were complete. Most significant assessment findings were related to inadequate corrective actions for a previously identified CR condition.

Nuclear Oversight findings with less significance contributed to an existing backlog of CR issues which was more pronounced on Unit 2 than on Unit 3. No significant differences in Nuclear Oversight assessment quality, number of findings, or finding significance were noted between units.

c. Conclusions

NNECO Nuclear Oversight adequately performed assessments of Engineering department related activities. Assessment activities and findings were appropriately oriented toward maintaining the design basis of the units, ensuring the operability of safety related equipment, identifying conditions of increased risk, and maintaining NNECO quality standards. Most Nuclear Oversight assessment findings were administrative and/or process related in nature and the most significant assessment findings were generally related to inadequate corrective actions for a previously identified CR condition.

E7.4 Engineering Self-Assessments

a. Inspection Scope (37550)

The team evaluated the effectiveness of Engineering department self-assessments in supporting the resolution of operating, technical and regulatory issues.

b. Observations

Based on a sample of Engineering department self assessments, the team determined that engineering self assessments were detailed, varied in scope, well supported, thoroughly referenced and safety oriented. Many of the self assessment findings were of a technical nature, but most were administrative. Assessment findings were assigned to the CR process, given appropriate CR significance, adequately tracked and managed, and corrected in a timely manner. Some low level Engineering department assessment findings have contributed to existing CR backlogs on both units. Based on the sample of Engineering Department assessments reviewed by the team, findings were typically of the CR significance level 2 and 3, with no immediate impact on the safe operations of the units.

c. Conclusion

Sampled NNECO Engineering department self assessments were detailed, varied in scope, well supported, thoroughly referenced and safety oriented. Many of the self assessment findings were of a technical nature, but most were administrative. Assessment findings were assigned to the CR process, given appropriate CR significance, adequately tracked and managed, and corrected in a timely manner. Typical self assessment issues were assigned CR significance levels of 2 or 3, and had no immediate impact on the safe operations of the units.

IV. MANAGEMENT MEETINGS

X1 Exit Meeting Summary

The results of the inspection were discussed with members of the engineering staff at a meeting on September 24, 1999. At that time, the inspection findings were not contested. Additional discussions were held on October 22, 1999 regarding the AFW turbine stem binding issue.

The inspection team did review several proprietary documents. Those documents identified as being proprietary were returned to NNECo at the end of the inspection.

X2 Management Meeting Summary

A meeting was held August 27, 1999, in the Region I office to describe the reorganization of the engineering department for the Millstone station, and discuss NNECo's strategy for improving performance in engineering. A brief discussion of the engineering backlog reduction program was also conducted. The NNECo slides used during the presentation are included as an Attachment to this report.

PARTIAL LIST OF PERSONS CONTACTED

B. Wilkins	Director, Design Engineering
G. Olsen	Manager, Millstone 3 Design Engineering
D. Smith	Manager, Regulatory Affairs
D. Dodson	Supervisor, Millstone 3 Licensing
B. Young	Supervisor, Design Engineering
D. Aube	Supervisor, Millstone 3 Instrumentation and Controls Design
G. Tardif	Supervisor, Engineering Assurance
D. Van Dyne	Supervisor, Civil/Structural Engineering
P. L'Heureux	Supervisor, Mechanical Systems Engineering
L. Salyards	Nuclear Oversight Auditor
B. Bohmbach	Instrumentation and Controls Engineer, Millstone 3
T. Cleary	NRC Coordinator
D. Harris	NRC Coordinator
L. Arzamarski	Licensing Assistant

INSPECTION PROCEDURES USED

IP37001	10 CFR 50.59 Evaluation Program
IP37550	Engineering

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

URI 50-423/99-10-01	Inconsistencies in calculation for required trisodium phosphate inventory in Unit 3 containment sumps
NCV 50-336&423/99-10-02	Failure to correct identified conditions adverse to quality

Closed

NCV 50-336&423/99-10-02	Failure to correct identified conditions adverse to quality

Discussed

None

LIST OF ACRONYMS USED

AFW	Auxiliary feedwater
CFR	Code of Federal Regulations
CR	Condition Report
DCR	Design Change Record
EWR	Engineering Work Request
FSAR	Final Safety Analysis Report
ISEG	Independent Safety Engineering Group
LOCA	Loss of Coolant Accident
MRT	Management Review Team
NNECo	Northeast Nuclear Energy Company
NUTIMS	Northeast Utilities Training Information Management System
OD	Operability Determination
OSM	Operations Shift Manager
PPM	Parts Per Million
RCS	Reactor Coolant System
SE	Safety Evaluation
SCTR	Steam Generator Tube Rupture
SPROC	Special Procedure
TM	Temporary Modification
TSP	Trisodium Phosphate

ATTACHMENT 1

List of Documents Requested from Northeast Nuclear Energy Company

List of Temporary Modifications currently open, as of September, 1999
List of Temporary Modifications restored within previous year
List of Permanent Modifications completed within previous year

Open Temporary Modifications

3-99-026, Installation of Check Valve 3SWP*V109 without internals
3-98-045, Control Building Service Water Pipe Chase Closure Plate Control Room Habitability Seal
3-96-069, 3SWP*V298 Repair per NCR 396-318

Restored Temporary Modifications

3-99-019, Defeat Automatic Closure of 3RCS*LCV460 in accordance with OP 3216
3-99-014, Temporary Power to 3BYS*PNL-2V
3-99-013, Temporary Power to 3BYS*PNL-1V
3-98-068, Leak Repair Valve 3FWS*CTV41B
3-98-063, Service Water Strainer Blowdown
3-97-026, Belzona® seat on 3SWP*TV35B

Permanent Modifications

EWR NO.	Document No.	Description
M3-97163	MSEE 0-1130-98	Modification to Service Water Pump Discharge Valve
M3-95-061A	MMOD M3-98039	Service Water MMOD 98039 - Additional Flanges - Task #7
M3-93020	PDCR 3-93-020	Grounding Stud Replacement on 4.16/9.6 kV Switch PMR 3-92-030-00
M2-94212	MSEE 0-1127-98	Bypass QSS-P1A/B Auto Trip on Low RWST Temperature
M3-93143	MSEE 0-0939-98	Level Switch Vibration Study
M3-95029	DCR M3-98047	Replacement of ‘D’ S/G AFW Check Valves and Addition of AFW Isolation Valves
M3-95302	DCR M3-99004	Aux Feedwater Pump Design Changes to 3FWA*P2

Plant Procedures
Design Control
Temporary Modifications
Safety Assessment
Work Control
Configuration Management
Independent Safety Engineering Group

Other Documents
Engineering Self-Assessments performed during the prior year
QA Audit reports of reviews of Engineering during the previous year
ISEG reviews performed during the prior year

ATTACHMENT 2

List of Documents Reviewed During Inspection

Nuclear Oversight Engineering Assessments

Unit 2

Engineering Periodic Assessment for December 1998
Engineering Periodic Assessment for January 1999
Engineering Periodic Assessment for February 1999
Engineering Periodic Assessment for March 1999
Engineering Periodic Assessment for April 1999
Engineering Periodic Assessment for May 1999
Engineering Periodic Assessment for June 1999
Engineering Periodic Assessment for July 1999
Nuclear Oversight Assessment Report, Calculation Assessment, dated April 6, 1999
Recovery Oversight Assessment of Unit 2 System Engineering Effectiveness, dated March 19, 1999
Recovery Oversight Assessment of Unit 2 Design Engineering Safety Evaluation Application, dated April 5, 1999
Recovery Oversight Assessment of Unit 2 Design Change Implementation, dated May 3, 1999
Independent Safety Engineering Report (AR 98020656, dated 2/9/99, Recovery Oversite Request MP2, TQ1, MEPL, NUTIMS

Unit 3

Nuclear Oversite Verification Plan - Conduct of Engineering, dated January 7, 1999
Nuclear Oversite Verification Plan - Conduct of Engineering, dated February 9, 1999
Nuclear Oversite Verification Plan - Conduct of Engineering, dated March 11, 1999
Nuclear Oversite Verification Plan - Conduct of Engineering, dated April 16, 1999
Nuclear Oversite Verification Plan - Conduct of Engineering, dated April 28, 1999
Nuclear Oversite Verification Plan - Conduct of Engineering, dated June10, 1999
Nuclear Oversite Verification Plan - Conduct of Engineering Training, dated September 10, 1999

MP-98-A20	Material Equipment and Parts Lists (MEPL)
MP-99-A03	Plant Modifications
MPS -ES-99-002	Plant Engineering Assessment for July 1999, dated 8/10/99
MPS -ES-99-003	Plant Engineering Assessment for July 1999, dated 8/12/99
MP-99-A12	Procurement
MP-99-A08	Special Nuclear/Licensed Materials
MP-99-A07	Process Control/Radwaste
MP-99-A01	Document Control and Quality Records
MP-99-A05	Special processes Programs (Welding, Non-destructive Examination, Coatings and Freeze Sealing)
MP-99-A06	Corrective Action Program
MP-99-A14	Unit 3 Refueling Activities
PES-98-040	Acceptance of Quality Calculations and Analyses

Operability Determinations

Unit 2

MP2-235-96	Emergency Diesel Generator Fuel Limit
MP2-255-96	Emergency Diesel Generator Bearing Failure
MP2-255-96	Emergency Diesel Generator Ventilation System Fan Test Failure
MP2-019-97	Emergency Diesel Generator Fuel Limit
MP2-037-98	Spent Fuel Pool Siphon Breakers
MP2-003-99	Loss of Normal Feedwater Accident Analysis
MP2-206-96	Emergency Diesel Generator Room Drains
MP2-021-98	Non-QA Parts in Safety Related Systems (MEPL)
MP2-029-98	Fire Pump Drain Line Check Valve
MP2-005-99	Reactor Building Penetration Leakage
MP2-045-99	4160 Switchgear Room Cooling
MP2-042-99	Charging Pump Line Crack

Unit 3

MP3-208-96	Component Cooling water Relief Valve Setpoints
MP3-209-96	Emergency Diesel Generator Logic
MP3-211-96	Emergency Diesel Generator Fuel Oil Delivery
MP3-028-98	Service Water Cubicle Ventilation
MP3-031-98	Recirculation Spray System Injection Pathway
MP3-036-96	Emergency Diesel Generator Thermal Performance Testing
MP3-039-98	Recirculation Cubicle Sumps
MP3-048-98	Non-Q Bearings Installed under Automatic Work Orders
MP3-019-99	Solenoid Failures
MP3-026-99	Recirculation Cubicle Sump Pump Test Failure
MP3-071-98	Main Steam Atmospheric Relief Isolation Valves
MP3-084-98	Hydrogen and Nitrogen Gas accumulation in the Charging System
MP3-004-99	Recirculation Spray Pump Mechanical Seal
MP3-011-99	Volume Control Tank Isolation Valves
MP3-030-99	Motor Operated Valves Missing T-Drains
MP3-003-99	Control Room Habitability

Special Procedures (SPROC) and Other Plant Procedures

Safety Injection System MOV Dynamic Test, SPROC MOV98-3-03, revision 0, dated 1/21/98
Boric Acid Batching Methods, SPROC EN98-3-07, revision 1, dated 5/8/98
Spent Fuel Pool Cooling System Vortex Suppressor, SPROC EN98-3-09, revision 0, dated 5/11/98
Immediate Boration, AOP 3566, revision 5 dated 6/14/98
Freeze Seal Repair of Unit 3 Reactor Coolant System Valve 3RCS-V132
Emergency Diesel Generator Voltage and Frequency Transient Monitoring, SPROC EN98-3-03, revision 1, dated 4/28/99

Spent Fuel Pool Vortex Suppressor Performance, SPROC EN98-3-14, revision 0, dated 5/30/98
Turbine Driven Auxiliary Feedwater Pump Operational Readiness Test, SP 3622.3

Millstone Unit 3, Operator Round Sheet 12, Revision 4 Change 4
Titration of Boric Acid with Trisodium Phosphate, dated November 3, 1994
Titration and Bulk Density of Trisodium Phosphate , dated March 21, 1995

Design Change Requests (DRS.), 10 CFR 50.59 Assessments and Associated DCR Summaries

Unit 2

M2-97026	Pressure Relief Modification for the Containment Sump Pump
M2-97025	Emergency Core Cooling Systems Pump Coolers Piping Modification

Unit 3

M3-971480	Actuator Gear Replacement for the Emergency Boration Bypass Isolation Valve
M3-98007	Reduced AFW Flow Rates to Support Final Safety Analysis Report, Chapter 15 Analysis
M3-98053	Main Steam Isolation Valve Solenoid Design Upgrade
M3-97109	Boric Acid Tank Alarm Setpoint Change
M3-97083	Setpoint Revision for Reactor Coolant Pump Underspeed Reactor Trip
M3-97026	Rerate of Auxiliary Feedwater System Inside Containment
M3-97095	4.16KV Feeder Circuit Fault Clearing Time
M3-96077	ECCS Orifice and Throttle Valve Balancing
M3-98028	Cycle 6 Reload Safety Evaluation
M3-98034	Setpoint Change for Pressure Control Valve 3GSN-PCV-106ECCS
M3-97219	RHR Suction Relief Valve Capacity
M3-97135	Filling Unit 3 Tri-sodium phosphate Baskets
M3-99004	Turbine Driven Auxiliary Feedwater Pump Stem and Rotating Assembly Replacement

Engineering Self Assessments

Unit 2

CM-SA-99-007	DCR Quality
M2-DE-99-002	Management Effectiveness
PES-97-032	Post Modification Test Plans
CM-SA-99-014	Engineering Support Training Accreditation

Unit 3

PES-SA-98-003	Engineering Plant Design Data System (PDDS) Relief Valve Setpoint Process
3DE-SA-99-02	Work Management Effectiveness
3DE-SA-99-05	Work Document Validation and Verification
DE-3-98-006	Controlled Manual Updates
DE-3-98-004	Temporary Modifications
DE-3-98-005	Personnel Qualifications
PES-SA-98-039	Engineering Attention to Quality
PES-SA-98-008	Temporary Modifications
M3-97-1217	Assessment of Twelve Electrical Protection Calculations

M3-97-0119	Assessment of Voltage Profile Analysis
CM-SA-99-006	Configuration Management Team
ENG-SA-99-001	Personnel Qualifications for Design Activities
ESAR-98-010	Engineering Work Requests (EWR) and Engineering Work Assignments (EWA) Validation and Verification
3DE-SA-98-08	Design Engineering Effectiveness
PES-SA-98-040	Acceptance of Purchased Quality Calculation and Analyses
ESAR-98-007	Design Engineering Personnel Qualification Records
ESAR-98-003	Review of FSAR Chapters 3 and 6 for Consistency
DE-SA-98-002	Review of Unit 3 Outage Modifications

Safety Evaluations and associated Calculations

Unit 2

S2-EV-98-0163	Insituform Service Water Liner Defects
S2-EV-98-0178	MEPL
S2-EV-96-018	Service Water Flow Calculations
S2-EV-98-0162	Engineering Functions

Unit 3

S3-EV-99-0009	Auxiliary Feedwater Pump Rotating Assembly and Governor Valve Stem Material Replacement
FSAR 99-MP3-92	Tri-sodium phosphate Storage
S3-EV-97-0323	Tri-sodium phosphate
USB-350 Revision b	Tri-sodium phosphate pH Calculations

Hydraulic Model Study of Millstone Unit 3 Spent Fuel Pool Cooling Vortexing, dated January 1, 1998

ISEG Reports

AR No. 98013465, Reactivity Management Events
AR No. 98011449, Organizational Communications
AR No. 98013467, Millstone Unit No. 3 Operator Work-Arounds
AR No. 98013456, Reactivity Management Events
AR No. 98013764.02, Millstone Unit 2 Refueling Preparations
AR No. 98013766, Use of Industry Operating Experience
AR No. 98019669, Feedwater Heater System Water Hammer Events
AR No. 98020656, Engineering Qualification
AR No. 99000535, Circulating Water Pumphouse Ventilation Fan 10A/B Failures
AR No. 99002025, Problem Identification
AR No. 99003018, NSE ISEG Walkdown of MP-3
AR No. 99004486, MP-3 Spent Fuel Pool Cooling Outage
AR No. 99005612, Spontaneous ISEG of MP-3 125 VDC Circuit Breaker Corrective Actions

AR No. 99007195, Preventing the Use of a Procedure in the Revision Phase
AR No. 9900850, Reactor Head Removal Pre-Job Brief
AR No. 99008109, MP-3 S/D Risk Management

References

NNECO References

TQ-1	Personnel Training Qualifications
RAC 5	10 CFR 50.72 Notifications
RP 4	Corrective Action Program
RP 5	Operability Determinations
RAC 12	Safety Evaluations
NGP 2.3 rev. 6	Differing Professional Opinions
NOQP 1.06 rev. 1	Nuclear Oversight Issues Resolution
NOQP 3.04 rev. 3	Independent Safety Engineering Group Operating Experience Assessment

Northeast Utilities Memo, Corrective Action Escalation Policy, dated 6/9/97

Other References

10 CFR 50, Appendix B Criterion III, Design Control
10 CFR 50, Appendix B Criterion XVII, Quality Assurance Records
10 CFR 50.59, Changes, Tests, and Experiments
10 CFR 50.2, Design Bases
ANSI N18.7-1976, Administrative Controls and Quality Assurance for the Operations Phase of Nuclear Power Plants
ANSI N45.2.9, Requirements for Collection, Storage, and Maintenance of Quality Assurance Records for Nuclear Power Plants
NRC Policy Statement, Availability and Adequacy of Design Bases Information at Nuclear Power Plants," August 10, 1992
SECY-91-364, Design Document Reconstitution

Condition Reports (CRs) and Adverse Condition Reports (ACRs)

Unit 2

M2-97-0275	Training Records - NUTIMS
M2-97-0679	Training Records - ANSI Standards
M2-98-1896	Training Records - NTM
M2-98-2900	Training Records - NUTIMS
M2-98-3029	Training Records - NUTIMS
M2-98-2357	Training Records - MEPL
M3-99-2575	Corrective Actions for the Resolution of Engineering Training Records
M2-98-2709	Secondary Water Chemistry Control
M2-99-1561	Special Nuclear Material Control Area Labeling

Unit 3

M3-98-0459	Tri-sodium phosphate Sump Concentrations
M3-012327	Tri-sodium phosphate Sump Concentrations
M3-99-2575	Corrective Actions for the Resolution of Engineering Training Records
M3-99-0974	Ineffective Corrective Actions for the Resolution of Organization Changes Related to the license Basis
M3-99-2233	Welder Qualification Records
M3-99-2507	Training Records
M3-99-2964	Security Officer Training Records
M3-99-0939	TQR Record Deficiency
M3-99-1200	Radiological Waste Worker Qualification
M3-99-0542	Ineffective Corrective Actions for the Resolution of Organization Changes Related to the License Basis
M3-99-0286	Ineffective Corrective Actions for the Control of Radiographs
M3-99-0646	Corrective Actions to Prevent Recurrence of Software Control Deficiencies.
M3-99-1511	Corrective Actions for Environmental Composite Discharge Samples
M3-99-1639	Corrective Actions for the Control of DCR Records
M3-98-4426	MEPL Deficiencies
M3-98-2357	Nuclear Oversight Training and Qualification
M3-99-3144	Preventive Maintenance Program
M3-98-2357	Training Records - MEPL
M3-99-0616	Training Records - Engineering Support